NON-PUBLIC?: N
ACCESSION #: 9412230211
LICENSEE EVENT REPORT (LER)

FACILITY NAME: Edwin I. Hatch Nuclear Plant - Unit 1 PAGE: 1 OF 8

DOCKET NUMBER: 05000321

TITLE: Main Turbine Shutdown Results in Automatic Reactor Shutdown

EVENT DATE: 11/19/94 LER #: 94-014-00 REPORT DATE: 12/16/94

OTHER FACILITIES INVOLVED: DOCKET NO: 05000

OPERATING MODE: 1 POWER LEVEL: 059

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR SECTION:
50.73(a)(2)(iv)

LICENSEE CONTACT FOR THIS LER:
NAME: Steven B. Tipps, Nuclear Safety TELEPHONE: (912) 367-7851
& Compliance Manager, Hatch

COMPONENT FAILURE DESCRIPTION:
CAUSE: SYSTEM: COMPONENT: MANUFACTURER:
REPORTABLE NPRDS:

SUPPLEMENTAL REPORT EXPECTED: NO

ABSTRACT:

On 11/19/94 at 0005 EST, Unit 1 was in the Run mode at a power level of 1437 CMWT (59 percent rated thermal power). Load had been reduced from about 99 percent rated thermal power 24 minutes earlier in response to a complete loss of feedwater heating. At 0005 EST, the reactor automatically shut down per design on Turbine Stop Valve closure. The Turbine Stop Valves closed when the turbine shut down on Moisture/ Separator Reheater (MSR) A/B high water level. Reactor vessel water level decreased from its normal value of 36 inches to a minimum of seven inches (151 inches above the top of the active fuel). The Group 2 Primary Containment Isolation System valves automatically closed on low reactor vessel water level per design. The Reactor Feedwater Pumps restored and maintained water level. No Emergency Core Cooling Systems were required to actuate. Reactor pressure was adequately controlled by

the Turbine Bypass valves.

The cause of the loss of feedwater heating was a damaged wire in a drain pot level switch combined with out-of-adjustment heater controls. This switch was repaired on 11-19-94. A more restrictive thermal limit was temporarily imposed on the core to insure a similar loss of feedwater heating event will not result in exceeding fuel design limits.

Investigation did not reveal any equipment problems that would have led to a high water level condition in the MSR.

END OF ABSTRACT

PLANT AND SYSTEM IDENTIFICATION

General Electric - Boiling Water Reactor
Energy Industry Identification System codes are identified in the text as (EIIS Code XX).

DESCRIPTION OF EVENT

On 11/18/94 at 2215 EST, Unit 1 was in the Run mode. At that time, Operations personnel reduced power to approximately 2217 CMWT (91 percent rated thermal power) to complete routine weekly control rod (EIIS Code AA) exercising per plant surveillance procedure 34SV-C11-003-1S, "Control Rod Weekly Exercise." At 2255 EST, Operations personnel satisfactorily completed procedure 34SV-C11-003-1S and began to increase power to 243 6 CMWT (100 percent rated thermal power).

At 2315 EST, Unit 1 was at about 99 percent rated thermal power and on a preconditioning ramp to 100 percent power. At that time, Operations personnel began routine testing of Main Turbine (EIIS Code TA) valves per plant test procedure 34IT-N30-001-1S, "Main Turbine & Auxiliaries Weekly Test." Testing consists of, one at a time, fully closing and then fully opening each valve. At 2340 EST, Operations personnel had successfully completed testing the four Turbine Stop Valves (EIIS Code TA) and three of the four Turbine Combined Intermediate Valves (CIVs, EIIS Code TA) without incident.

At 2340 EST, as Operations personnel were opening CIV #4, the last turbine valve to be tested, numerous low pressure feedwater heater alarms were received in the Main Control Room. Reactor power increased slightly and the low pressure feedwater heaters (EIIS Code SJ) extraction steam line pressures spiked high. Based upon the alarms and indications of unexpected power and extraction steam line pressure changes, Operations personnel concluded that a loss-of-feedwater-heating event had occurred and entered abnormal operating procedure 34AB-N21-001-1S, "Loss of Feedwater Heating." They immediately reduced power by 20 percent rated thermal power, as required by the procedure, by reducing Recirculation System (EIIS Code AD) pump flow rate. Final feedwater temperature, as measured at the outlet from the high pressure (5th stage) feedwater heaters (EIIS Code SJ), began to decrease due to the decrease in reactor power and the isolation of the feedwater heaters. Final feedwater temperature dropped from its pre-event value of approximately 383 degrees F to a low of about 227 degrees F within five minutes. Of this 156 degrees F drop in temperature, about 110 degrees F was attributed to the loss of feedwater heating and 46 degrees F was attributed to the reduction in reactor power.

Operations personnel further reduced power to 70 percent rated thermal power by inserting some control rods. Main Control Room and local feedwater heater control panel indications showed some of the feedwater heaters were returning to service (the heaters automatically return to service when the condition that caused their isolation has cleared). Final feedwater temperature began to increase and, at about 2355 EST, had returned to 330 degrees F, further reducing reactor power to 59 percent. The feedwater temperature of 330 degrees F was approximately 11 degrees F below normal for 59 percent power when all feedwater heaters are in service, but was within the acceptable region of operation per the feedwater temperature versus core power graph found in plant operations procedure 34GO-OPS-005-1S, "Power Changes." Feedwater temperature and reactor power stabilized at 333 degrees F and 59 percent, respectively, and remained constant for about ten minutes.

At 0005 EST, on 11/19/94, Unit 1 was in the Run mode at a power level of 1437 CMWT (59 percent rated thermal power). At that time, annunciator "MSR A/B Tank Level High" alarmed. This indicated the water level in the Moisture/Separator Reheater (MSR, EIIS Code SB) A/ B drain tank (EIIS Code SN) was unusually high. Approximately 30 seconds after receipt of this alarm, annunciator "MSR High Level Trip" alarmed. The turbine shut down and the reactor automatically shut down per design on Turbine Stop Valve closure. Reactor vessel water level decreased from its normal value of 36 inches above instrument zero to a minimum of seven inches below instrument zero (151 inches above the top of the active fuel) due to void collapse from the rapid reduction in reactor power. The Group 2 Primary Containment Isolation System (EIIS Code JM) valves closed and another automatic reactor shutdown signal was received on low reactor vessel water level per design. The Reactor Feedwater Pumps (EIIS Code

SJ) automatically restored water level within 30 seconds of the reactor shutdown and were used to maintain water level thereafter. No Emergency Core Cooling Systems actuated nor were any required to actuate to recover or maintain water level.

Reactor vessel pressure increased to a peak of 1020 psig from its pre-event value of 956 psig. The Turbine Bypass Valves (EIIS Code SO) opened automatically to lower and maintain reactor pressure. No Safety/Relief Valves opened nor were any required to open to reduce or control pressure.

CAUSE OF EVENT

The apparent cause of the loss-of-feedwater-heater event was a damaged wire in drain pot level switch 1N36-N064B for the "B" 10th stage feedwater heater extraction steam line (EIIS Code SE). The insulation on the wire was damaged resulting in a portion of the wire being exposed; the exposed portion of the wire grounded to the level switch housing. The ground caused breaker 38 in

Distribution Panel 1R25-S021 to open on high current. This, in turn, caused the 10th and 12th stage "N" and "B" low pressure feedwater heaters to isolate; that is, their extraction steam line isolation valves closed and the low water level heater control valves from the next higher pressure heater also closed. The feedwater heater system transient caused by the sudden isolation of the 10th and 12th stage "A" and "B" feedwater heaters apparently resulted in the 7th and 8th stage low pressure heaters, and possibly the 5th stage "A" and "B" high pressure heaters, to isolate on high heater water level.

A heater isolation on high water level will cause isolation of the respective heater extraction steam line and closure of the low water level control valve for the next higher pressure heater, resulting in loss of heating for the feedwater passing through the heater. Consequently, isolation of some or all of the feedwater heaters will cause a drop in final feedwater temperature. From the magnitude and rate of temperature drop, it appears all low pressure and high pressure feedwater heaters were isolated for a short period of time (less than 17 minutes). It also appears from the subsequent feedwater temperature increase that all the feedwater heaters except the 10th and 12th stage "A" and "B" low pressure heaters automatically returned to service as water levels decreased through their high water level control valves. The 10th and 12th stage heaters did not return to service because the open breaker prevented their extraction steam line isolation and low water level control valves from opening.

Ideally, high level dump valves from each feedwater heater will open to route water from the heater to the main condenser in the event of a high level in the feedwater heater. These valves should have opened to prevent the high level isolation which occurred on the 8th, 7th, and 5th stage heaters. However, with feedwater heater levels changing rapidly, the heater controls may not have been adjusted to allow the high level dump valves to operate in time to prevent the high-high level isolation.

The cause of the high water level in MSR A/B could not be determined. Because the MSR high water level must be sensed by two out of three level switches in order to shut down the turbine, it appears a true high water level condition existed in MSR A/B. However, investigation did not reveal any problems that would have led to a high water level in the MSR. Inspection of MSR A/B water level switches following the automatic reactor shutdown showed all three had reset; this was expected since the water in the MSR should have drained following the shutdown of the turbine. Therefore, it appears the switches were functioning properly. Testing of MSR A/B drain tank low and high water level control valves and their associated controllers revealed no problems that would have caused them to fail to maintain MSR A/B water level below the high level setpoint. No blockages in either the low level or high level drain lines were found or evident.

The volume booster air supply isolation valve for the MSR A/B drain tank high water level control valve actuator was found closed following the automatic reactor shutdown. The air supply isolation valve should have been open because the volume booster provides another source of air, in addition to the normal air supply, to the control valve actuator. The purpose of the volume booster is to
lose the high water level control
valve more quickly on a decreasing water level signal from the valve controller. It also provides a larger vent area to allow the valve to open more quickly on an increasing water level signal. (This control valve is an air-to-close, spring-to-open valve.) The valve was required to be closed by plant procedure 34SO-P51-002-1S, "Instrument and Service Air Systems," because it was shown incorrectly as normally closed on a plant piping and instrumentation drawing; however, the fact that it was closed had no adverse effect on the ability of the control valve to open. The control valve should have gone to the fully open position in response to the high water level condition in MSR A/B drain tank. The closed volume booster air supply isolation valve did not affect the ability of the volume booster to vent the valve actuator and allow the valve to open

quickly. Consequently, the closed volume booster air supply isolation valve did not cause the high water level condition in MSR A/B.

Although it seems likely the loss-of-feedwater-heating event contributed to MSR A/B high water level, no connection between the two events could be established. In fact, it was about ten minutes after the 5th, 7th, and 8th stage "A" and "B" feedwater heaters returned to service and feedwater temperature stabilized when the turbine shut down on MSR A/B high water level,

REPORTABILITY ANALYSIS AND SAFETY ASSESSMENT

This report is required by 10 CFR 50.73(a)(2)(iv) since an unexpected actuation of the Reactor Protection System (EIIS Code JC) and the Group 2 Primary Containment Isolation System, Engineered Safety Feature systems, occurred. Specifically, MSR A/B high water level resulted in an automatic shutdown of the turbine per design. The Turbine Stop Valves close to shut down the turbine; Turbine Stop Valve closure is a Reactor Protection System actuation signal. The reactor therefore automatically shut down per design. The decrease in reactor water level caused by the sudden decrease in reactor power resulted in isolation of the Group 2 Primary Containment Isolation System valves on low reactor vessel water level.

The Moisture Separator/Reheaters, located between the high and low pressure turbines, remove moisture from, and reheat, the exhaust steam from the high pressure turbine before it goes into the low pressure turbines. This improves the quality of the steam going to the low pressure turbines thereby reducing turbine damage from moisture in the steam. The moisture separated from the steam traveling through the MSRs is routed to two MSR drain tanks, one for each set of two MSRs. The water in the drain tanks is, in turn, routed to the 7th stage "A" and "B" low pressure feedwater

heaters where its energy is used to heat the feedwater prior to its reaching the reactor pressure vessel. If the 7th stage heaters are isolated, then the drain lines from the MSR drain tanks to the heaters are isolated and the water in the MSR drain tank is routed to the Main Condenser (EIIS Code SQ) through high water level control drain lines.

If the water level in the MSR drain tanks gets too high, water will back up into the MSRs where it can back up into the turbine. Water intrusion can cause major damage; therefore, the turbine is shut down on MSR high water level prior to water reaching the point of intrusion. The turbine is shut down by closure of the Turbine Stop and Control Valves which isolates its source of motive steam.

Closure of the Turbine Stop and Control Valves will produce reactor pressure, neutron flux, and heat flux transients. Therefore, automatic reactor shutdown is initiated on Turbine Stop and Control Valve closure signals to reduce the amount of energy in the reactor pressure vessel and, hence, to limit the size of the accompanying pressure and flux transients.

In this event, the turbine shut down per design when water level in MSR A/B increased to the high level setpoint. This prevented water induction damage to the turbine. The shutdown of the turbine resulted in an automatic reactor shutdown per design. The Group 2 Primary Containment Isolation System valves automatically closed on low reactor vessel water level per design. All systems functioned properly to limit the reactor vessel water level and pressure transients. Water level dropped only 43 inches before the Reactor Feedwater Pumps automatically restored it to normal levels. At no time was water level less than 151 inches above the top of the active fuel. The Turbine Bypass Valves opened automatically to limit the pressure increase to 64 psig. No Emergency Core Cooling Systems were required to control water level nor were any Safety/Relief Valves required to control pressure.

Prior to the automatic reactor shutdown, a loss-of-feedwater-heating event occurred which resulted in a drop in final feedwater temperature of more than 100 degrees F. The magnitude of the temperature drop exceeded the maximum 100 degrees F drop from a single failure assumed in Unit 1 Final Safety Analysis Report section 14.3.2.2. Therefore, this event was analyzed for its effects on the fuel and fuel thermal limits. The Southern Nuclear Fuels Group reviewed the event and determined that, given the fuel thermal limits margin that existed prior to the event and the power reduction that occurred during the event, sufficient margin existed to preclude any thermal limits violations during the feedwater temperature transient which resulted from the loss-of-feedwater-heating event.

Based upon the preceding analysis, it is concluded that this event had no adverse effect upon nuclear safety. This analysis is applicable only to this event.

CORRECTIVE ACTIONS

The wire in drain pot level switch 1N36-N064B was repaired on 11/19/94 and breaker 38 in Distribution Panel 1R25-S021 was reset. Turbine valve testing was performed successfully on 12/2/94 with no incidents or unusual occurrences.

Prior to reactor startup, the Southern Nuclear Fuels Group reviewed the loss-of-feedwater-heating event in which final feedwater temperature decreased by greater than 100 degrees F. They determined that this event had no adverse impact on the fuel. Given the fuel thermal limits margin that existed prior to the event and the power reduction that occurred during the event, they determined that sufficient margin existed to preclude any thermal limits violations during the feedwater temperature transient which resulted from the loss-of-feedwater-heating event.

As a conservative action, until such time as a determination is made that the performance of the feedwater heaters relative to the high level isolation is consistent with design assumptions, a MAPFACp curve (more restrictive than the one in the Unit 1 Core Operating Limits Report) is being implemented in the Unit 1 process computer. This limit will provide assurance that a loss of feedwater heating event in excess of 100 degrees F will not result in exceeding any Specified Acceptable Fuel Design Limits (SAFDLs).

The Architect/Engineer for Plant Hatch provided an evaluation of the design of the feedwater heater control system relative to its susceptibility to a single failure. The evaluation showed that the system should respond by diverting the excess water to the main condenser to prevent the isolation on high-high level; thus, the event did not expose a single failure vulnerability on the feedwater heater control system.

Heater controls will be adjusted in an attempt to improve the response of the system to a high level condition.

Procedure 34SO-P51-002-1S has been changed to require the air supply isolation valve to the volume booster to be open. The piping and instrumentation drawing will be revised by January 31, 1995, to show the valve as normally open.

ADDITIONAL INFORMATION

No systems other than those mentioned in this report were affected by this event.

No failed components caused or resulted from this event. The wire in drain pot level switch 1N36-N064B did not fail; the insulation on the wire appeared to have been damaged at some indeterminate time.

One previous similar event in the last two years in which the reactor automatically shut down on a turbine shutdown was reported in Licensee Event Report 50-366/1992-026, dated 12/21/92. In the previous event, the turbine shut down on high vibration that was not noted during a power increase. The vibration was the result of increasing power and was not related to or caused in any way by high water levels in the MSRs. Since the causes of the two events were different, the corrective actions for the previous event would not have addressed the causes of this event and, therefore, could not have prevented its occurrence.

ATTACHMENT TO 9412230211 PAGE 1 OF 1

Georgia Power Company
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Georgia Power

J. T. Beckham, Jr. the southern electric system
Vice President - Nuclear
Hatch Project December 16, 1994

Docket No. 50-321 HL-4755

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555

Edwin I. Hatch Nuclear Plant - Unit 1
Licensee Event Report
Main Turbine Shutdown Results
In Automatic Reactor Shutdown

Gentlemen:

In accordance with the requirements of 10 CFR 50.73(a)(2)(iv), Georgia Power Company is submitting the enclosed Licensee Event Report (LER) concerning a main turbine shutdown that resulted in an automatic reactor shutdown.

Sincerely,

J. T. Beckham, Jr.
OCV/et

Enclosure: LER 50-321/1994-014

cc: Georgia Power Company
Mr. H. L. Sumner, General Manager - Nuclear Plant
NORMS

U.S. Nuclear Regulatory Commission, Washington, D.C.
Mr. K. Jabbour, Licensing Project Manager - Hatch

U.S. Nuclear Regulatory Commission, Region II
Mr. S. D. Ebneter, Regional Administrator
Mr. B. L. Holbrook, Senior Resident Inspector - Hatch

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